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OMB Control No.: 3150-0012

UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR REACTOR REGULATION
WASHINGTON, DC 20555-0001

xxxxxxx, 2002

NRC BULLETIN 2002-xx: REACTOR PRESSURE VESSEL HEAD AND VESSEL HEAD PENETRATIONS INSPECTION PROGRAMS

Addressees

All holders of operating licenses for pressurized-water nuclear power reactors, except those who have ceased operations and have certified that fuel has been permanently removed from the reactor vessel, and all holders of operating licenses for boiling-water reactors for information.

Purpose

The U.S. Nuclear Regulatory Commission (NRC) is issuing this bulletin to:

(1) Advise pressurized-water reactor (PWR) addressees that visual examinations, as a primary inspection method for the reactor pressure vessel (RPV) head and vessel head penetrations (VHPs), may need to be supplemented with additional measures (e.g., volumetric and surface examinations) to demonstrate compliance with applicable regulations.

(2) Advise PWR addressees that inspection methods and frequencies to demonstrate compliance with applicable regulations should be demonstrated to be reliable and effective.

(3) Request information from all PWR addressees concerning their RPV head and VHP inspection programs to ensure compliance with applicable regulatory requirements.

(4) Require all PWR addressees to provide written responses to this bulletin related to their inspection program plans.

Background

Primary water stress corrosion cracking (PWSCC) in PWR control rod drive mechanism (CRDM) nozzles and other vessel head penetration nozzles fabricated from Alloy 600 is not a new issue; axial cracking in the CRDM nozzles has been identified since the late 1980s. In addition, numerous small-bore Alloy 600 nozzles and pressurizer heater sleeves have experienced leaks attributable to PWSCC. The area of interest for potential cracking of RPV head penetrations is the pressure-retaining boundary, which includes the J-groove weld between the nozzle and reactor vessel head and the portion of the nozzle inside the head.

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After reviewing safety assessments submitted by the industry and examining international inspection findings, the NRC staff concluded, in 1993, that CRDM nozzle and weld cracking, observed at that time, in PWRs was not an immediate safety concern. The basis for this conclusion was that if PWSCC occurred (1) the cracks would be predominantly axial in orientation, (2) the axial cracks would result in detectable leakage before catastrophic failure (with the expectation that CRDM nozzle cracking would result in a substantial volume of leaking coolant), and (3) the leakage would be detected during visual examinations performed as part of surveillance walkdown inspections before significant damage to the RPV head occurred. The safety evaluation identified concerns about potential circumferential cracking (which would need to be addressed on a plant-by-plant basis) as a consequence of high residual stresses resulting from initial manufacture and the impact of tube straightening that may have been needed after welding. The safety evaluation also noted the need for enhanced leakage monitoring.

On April 1, 1997, the NRC issued Generic Letter (GL) 97-01, "Degradation of Control Rod Drive Mechanism Nozzle and Other Vessel Closure Head Penetrations." Licensees' responses to GL 97-01 were predicated on development of susceptibility ranking models to relate the operating conditions (in particular the operating temperature and time) for each plant to the plant's relative susceptibility to PWSCC. The responses committed to surface examinations (i.e., eddy current) of the VHP nozzles at the plants identified as having the highest relative susceptibility ranking. The surface examinations conducted prior to November 2000 identified only limited axial cracking and circumferential cracking below the weld in the base metal of CRDM nozzles, but no circumferential cracking above the nozzle welds and no cracking in the Alloy 82/182 welds.

Inspections of the reactor nozzles at Oconee Nuclear Stations 2 and 3 in early 2001 identified circumferential cracking of the nozzles above the J-groove weld. Circumferential cracking above the J-groove weld is considered a safety concern because of the possibility of nozzle ejection should the circumferential cracking not be detected and corrected. On August 3, 2001, the NRC issued Bulletin 2001-01, "Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles" (ADAMS Accession Number ML012080284). The bulletin described instances of cracked and leaking Alloy 600 VHP nozzles, including CRDM and thermocouple nozzles, at Oconee Nuclear Station 3. In response to the bulletin, PWR licensees provided their plans for inspecting their VHP nozzles and/or the outside surface of the RPV head to determine whether the nozzles were leaking. Most plants have completed these inspections. Also, PWR licensees provided information on past leakage or cracking of VHP nozzles, RPV insulation type and configuration, and their susceptibility ranking.

During inspections of VHP nozzles initiated by NRC Bulletin 2001-01 in early March 2002, Davis-Besse Nuclear Power Station identified a large cavity in the RPV head near the top of the dome. This cavity was adjacent to a nozzle, which was leaking as a result of through-wall axial cracking, in an area of the RPV head that the licensee had left covered with boric acid deposits for a number of years. On March 18, 2002, the NRC issued Bulletin 2002-01, "Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity" (ADAMS Accession Number ML020770497). ~~Bulletin 2002-01 describes an instance of severe material degradation of the RPV head at Davis-Besse. In response to the bulletin,~~ The bulletin requested PWR licensees to provide~~d~~ information on RPV head inspection and maintenance programs, the material condition of ~~the~~ their RPV head, past incidents of boric acid leakage that could have reached the RPV head, and the basis for concluding that the boric acid inspection programs for the rest of the reactor coolant pressure boundary are effective. In their responses, they provided information on the extent to which they could conclude that they did not have RPV head degradation like that identified at Davis Besse.

Additional Information on Cracking and Material Degradation

The NRC has developed Web pages to keep the public informed of generic activities related to Alloy 600 cracking and RPV head degradation:

http://www.nrc.gov/reactors/operating/ops-experience/alloy600.html
http://www.nrc.gov/reactors/operating/ops-experience/vessel-head-degradation.html

These Web pages provide links to information regarding the nozzle cracking and head degradation identified to date, along with documentation of the NRC's interactions with industry (e.g., industry submittals, meeting notices, presentation materials, and meeting summaries). The NRC will continue to update these Web pages as new information becomes available.

Discussion

~~The NRC staff used licensees' responses to NRC Bulletins 2001-01 and 2002-01 to determine the need for, and guide the development of, additional regulatory actions to address cracking in VHP nozzles and the material condition of the RPV head and the rest of the reactor coolant pressure boundary.~~ As a result of the circumferential cracking of VHP nozzles at Oconee Nuclear Station 3 and other PWR facilities, the RPV head material degradation at Davis-Besse, and the staff's review of responses to NRC Bulletins 2001-01 and 2002-01, the NRC staff has a number of concerns about the inspection requirements and programs for RPV head and VHP nozzles. Based on the experience and information currently available concerning cracking and degradation, it may be necessary for inspection programs that rely on visual examinations to be supplemented with additional measures (e.g., volumetric and surface examinations) to demonstrate compliance with applicable regulations.

Undetected circumferential cracking of VHP nozzles and degradation of the RPV head can pose a safety concern if permitted to progress to the point that the integrity of the reactor coolant pressure boundary is in question and the probability of a loss-of-coolant accident (LOCA) or a VHP nozzle ejection increases. The discoveries of circumferential cracking of VHP nozzles and RPV head material degradation have raised several issues that prompted the staff to question the adequacy of current RPV head and VHP inspection programs that rely on visual examinations as the primary inspection method:

- Circumferential cracking of CRDM nozzles was identified by the presence of relatively small amounts of boric acid deposits. This finding increases the need for more effective visual and non-visual NDE inspection methods to detect the presence of degradation in CRDM nozzles before nozzle integrity is compromised.

- Cracking of Alloy 82/182 weld metal has been identified in CRDM nozzle J-groove welds for the first time and can precede cracking of the base metal. This finding raises concerns because examination of weld material is more difficult than base material. ~~regarding the adequacy of cracking susceptibility models based solely on base metal conditions~~.

- Through-wall circumferential cracking from the outside diameter of the CRDM nozzle has been identified for the first time. This raises concerns about the potential for failure of CRDM nozzles and control rod ejection, causing a LOCA.

- The environment in the CRDM housing/RPV head annulus will likely be more aggressive after any through-wall leakage because potentially highly concentrated borated primary water may become oxygenated. This raises concerns about the technical basis for current crack growth rate models.

- The presence of boron deposits or residue on the RPV head, due to leakage from mechanical joints, could mask pressure boundary leakage. This raises concerns that a through-wall crack may go undetected for years.

- The causative conditions surrounding the degradation of the RPV head at Davis-Besse have not been definitively determined. ~~This lack of certainty raises concerns whether material wastage (corrosion) rates and the mechanisms needed to cause material wastage have been appropriately considered in the selection of inspection methods and frequencies.~~ The staff is unaware of any data applicable to the geometries of interest that support accurate predictions of corrosion mechanisms and rates.

In summary, the discoveries of cracked and leaking Alloy 600 VHP nozzles at several PWRs and the RPV head degradation at Davis-Besse have raised concerns about the adequacy of current inspection programs that rely on visual examinations as the primary inspection method to ensure RPV head and VHP structural integrity and compliance with applicable regulations. Specifically, the staff is concerned that the inspection methods and frequencies (i.e., inspection intervals) of current inspection programs may not be sufficient. Based on the experience and information currently available, it may be necessary for inspection programs that rely on visual examinations to be supplemented with additional measures (e.g., volumetric and surface examinations) to demonstrate compliance with applicable regulations.

~~Currently, there are a number of activities by various organizations to address the questions and concerns stated in this bulletin. The staff expects that the results from the industry's ongoing and planned research effort, the industry's nondestructive examination demonstration program, and the NRC's confirmatory research programs will form the technical basis for a more effective inspection program for reactor vessel head and reactor vessel head penetrations. The staff plans to participate in ongoing efforts to revise the American Society of Mechanical Engineers (ASME) standard that governs RPV head and VHP inspections.~~

Issuance of this bulletin is the first step in a multi-step approach to address concerns about the adequacy of inspection requirements and programs for RPV heads and VHP ~~inspections~~. The other steps are: review the bulletin responses ~~to~~ and determine ~~if~~ what further ~~interim~~ regulatory actions are needed (i.e., revision to 10 CFR 50.55a), ~~continue to~~ review the Electric Power Research Institutes Material Reliability Program's (MRP's) proposed inspection program once an applicable technical basis is provided, ~~participate in~~ encourage the revision of American Society of Mechanical Engineers (ASME) inspection requirements, and, if acceptable, incorporate the revised ASME ~~standard~~ requirements into NRC regulations.

Example of Supplemental Inspections

Until a better understanding of the wastage phenomena and wastage rates has been developed, it may be necessary for inspection programs that rely on visual examinations to be supplemented with additional measures (e.g., volumetric and surface examinations). Table 1 provides an example of what the staff considers to be a reasonable set of supplemental inspections, based on current experience and understanding of material degradation and wastage rates. ~~gained to date and information currently available.~~

Table 1: Example of Reasonable Supplemental Inspections

Inspections	**Frequency/Time (Notes 1 and 2)**		
	< 8 EDY	**≥8 EDY and ≤12 EDY**	**> 12 EDY**
100% Ultrasonic Testing of CRDM Nozzle Base Material (Note 3) ***and***	within 5 years, then at least once every 60 full power months	every other refueling outage (not to exceed 48 full power months), beginning with the refueling outage after the next refueling outage	every refueling outage (not to exceed 24 full power months), beginning with the next refueling outage
100% Eddy Current Testing or Dye Penetrant Testing of all J-Groove Weld and CRDM Penetration Material Wetted Surfaces (Note 4) ***and***	within 5 years, then at least once every 60 full power months	every other refueling outage (not to exceed 48 full power months), beginning with the refueling outage after the next refueling outage	every refueling outage (not to exceed 24 full power months), beginning with the next refueling outage
100% Bare Metal Visuals Examination of CRDM to RPV Junction at Top of RPV Head (Note 5)	within 3 years, then at least once every 60 full power months	every other refueling outage (not to exceed 48 full power months), beginning with the next refueling outage	every refueling outage (not to exceed 24 full power months), beginning with the next refueling outage

Note 1: An effective degradation year (EDY) is a means for assessing the potential for cracking at a plant It accounts for the amount of time a plant has operated and the temperatures at which it has operated

Note 2. If a part through-wall flaw is identified in a plant with less than 8 EDY, then the guidance in the middle column becomes applicable Regardless of EDY, if through-wall or through-weld cracking is identified during the inspection, then the guidance in the last column becomes immediately applicable

Note 3: Testing should include as a minimum, the portion of the nozzle inside the RPV head to the bottom of the nozzle

Note 4: If ultrasonic testing has been demonstrated as reliable and effective in detecting and characterizing flaws in the J-groove weld, it may be used for inspections of J-groove welds

Note 5 If boron deposits or other indications of leakage are identified, then non-visual examination needs to be used to make a determine whether the leakage is from a through-wall or through-weld crack

Applicable Regulatory Requirements

Several provisions of the NRC's regulations and plant operating licenses (Technical Specifications) pertain to the issue of VHP nozzle cracking. Plant technical specifications pertain to the issue of VHP nozzle cracking insofar as they do not allow operation with through-wall reactor coolant pressure boundary leakage. The general design criteria (GDC) for nuclear power plants (Appendix A to 10 CFR Part 50), or, as appropriate, similar requirements in the licensing basis for a reactor facility, the requirements of 10 CFR 50.55a, and the quality assurance criteria of Appendix B to 10 CFR Part 50 provide the bases and requirements for NRC staff assessment of the potential for and consequences of VHP nozzle cracking and degradation of the reactor coolant pressure boundary.

~~The applicable design criteria include GDC 14 (Reactor Coolant Pressure Boundary) and GDC 31 (Fracture Prevention of Reactor Coolant Pressure Boundary). GDC 14 specifies that the reactor coolant pressure boundary (RCPB) have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture; the presence of cracked and leaking VHP nozzles is not consistent with this GDC. GDC 31 specifies that the probability of rapidly propagating fracture of the RCPB be minimized; the presence of cracked and leaking VHP nozzles is not consistent with this GDC.~~

NRC regulations in 10 CFR 50.55a state that ASME Class 1 components (which include VHP nozzles) must meet the requirements of Section XI of the ASME Boiler and Pressure Vessel Code. Various portions of the ASME Code address reactor coolant pressure boundary inspection. For example, Table IWB-2500-1 of Section XI of the ASME Code provides examination requirements for RPV head pressure retaining components and references IWB-3522 for acceptance standards. IWB-3522.1(c), (d), and (e) specify that conditions requiring correction include the detection of leakage from insulated components and discoloration or accumulated residues on the surfaces of components, insulation, or floor areas which may reveal evidence of borated water leakage, with leakage defined as "the through-wall leakage that penetrates the pressure retaining membrane." Even though the NRC is currently questioning the inspection requirements in the ASME Code, it is clear that the ASME Code, does not permit continued operation with through-wall degradation of the reactor pressure vessel head. Therefore, 10 CFR 50.55a, through its reference to the ASME Code, does not permit continued operation with through-wall degradation of the reactor pressure vessel head penetration nozzles.

Criterion V (Instructions, Procedures, and Drawings) of Appendix B to 10 CFR Part 50 states ~~that activities affecting quality shall be prescribed by documented instructions, procedures, or drawings, of a type appropriate to the circumstances and shall be accomplished in accordance with these instructions, procedures, or drawings. Criterion V further states~~ that instructions, procedures, or drawings shall include appropriate quantitative or qualitative acceptance criteria for determining that important activities have been satisfactorily accomplished. Visual, volumetric, and surface examinations of the reactor coolant pressure boundary are activities that should be documented in accordance with these requirements.

Criterion IX (Control of Special Processes) of Appendix B to 10 CFR Part 50 states that special processes, including nondestructive testing, shall be controlled and accomplished by qualified personnel using qualified procedures in accordance with applicable codes, standards, specifications, criteria, and other special requirements. Within the context of providing assurance of the structural integrity of the reactor coolant pressure boundary for the degradation observed at Davis-Besse, special requirements for visual examination and/or ultrasonic testing would generally require the use of qualified visual and ultrasonic testing methods. Such methods are ones that a plant-specific analysis has demonstrated would result in the reliable detection of degradation prior to a loss of specified reactor coolant pressure boundary integrity and margins of safety. The analysis would have to consider, for example, the as-built configuration of the system and the capability to reliably detect and accurately characterize flaws or degradation, and contributing factors such as access to the inspection area, the presence of insulation, preexisting deposits, and other factors that could interfere with the detection of degradation.

Criterion XVI (Corrective Action) of Appendix B to 10 CFR Part 50 states that measures shall be established to assure that conditions adverse to quality are promptly identified and corrected. For significant conditions adverse to quality, the measures taken shall include root cause determination and corrective action to preclude repetition of the adverse conditions. For cracking of VHP nozzles or material wastage of the RPV head, the root cause determination is important to understanding the nature of the degradation present and the required actions to mitigate future cracking or material wastage. These actions could include proactive inspections, repair of leaking VHP nozzles, and valid acceptance by analytical evaluation for degraded VHP nozzles where through-wall leakage may not be imminent.

<u>Requested Information</u>

The purpose of the information request is not to collect the same information that was submitted by PWR licensees in response to NRC Bulletins 2001-01 and 2002-01. The scope of this bulletin is broader than Bulletin 2001-01 because it addresses both material wastage and cracking, but the scope is narrower than Bulletin 2002-01 because it only addresses RPV head and VHP nozzles - not the entire pressure boundary. During the review of PWR licensees' responses to NRC Bulletins 2001-01 and 2002-01 and recent public meetings with NEI and MRP, a number of concerns have been raised about current inspection requirements and programs for RPV heads and VHP nozzles. The purpose of this bulletin is to learn what changes, if any, PWR licensees have made to their inspection programs for the RPV head and reactor VHP nozzles and their justification for reliance on visual examinations it that is their primary method to detect degradation.

(1) Within 30 days of the date of this bulletin:

A. PWR addressees who plan to supplement their ~~RPV head and VHP~~ inspection programs with non-visual NDE methods, are requested to provide a summary discussion (i.e., methods, EDY, scope, coverage, frequencies, qualification requirements, and acceptance criteria) of the supplemental inspections to be implemented.

B. PWR addressees who do not plan to supplement their inspection programs for ~~RPV head and VHPs~~ with non-visual NDE methods, are requested to provide a justification for continued reliance on visual examinations as the primary method to detect degradation (e.g., cracking, leakage, or wastage). In your justification, include a discussion that addresses the reliability and effectiveness of the inspections to ensure that all regulatory and technical specification requirements are met during the operating cycle, and that addresses the six concerns bulletized in the Discussion Section of this bulletin. Also, include in your justification a discussion of your basis for concluding that unacceptable vessel head wastage will not occur between inspection cycles that rely on qualified visual inspections. You should provide all applicable data to support your understanding of the wastage phenomena and wastage rates.

(2) Within 30 days after plant restart following the next inspection of the RPV head and VHP nozzles to identify the presence of any degradation, all PWR addressees are requested to provide:

A. the inspection scope and results, including the location, size, extent, and nature of any degradation (e.g., cracking, leakage, and wastage) detected; details of the NDE used (i.e., method, number, type, and frequency of transducers or transducer packages, essential variables, equipment, procedure and personnel qualification requirements, including personnel pass/fail criteria); and criteria used to determine whether an indication, "shadow," or "backwall anomaly" is acceptable or rejectable.

B. the corrective actions taken and the root cause determinations for any degradation found.

Required Response

In accordance with 10 CFR 50.54(f), each PWR addressee is required to submit a written response as described below. This information is sought to verify licensees' compliance with the current licensing bases for the PWR facilities covered by this bulletin.

Within 15 days of the date of this bulletin, ~~each~~ PWR addressees ~~is~~ are required to submit a written response ~~indicating (1) whether the requested information will be submitted and (2) whether the requested information will be submitted within the requested time period~~ if they are unable or choose not to provide the information or they can not meet the requested completion dates. PWR addressees who choose not to submit the requested information, or are unable to satisfy the requested completion date, must describe in their response any alternative course of action they propose, including the basis for the acceptability of the proposed alternative course of action.

The required written responses should be addressed to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, 11555 Rockville Pike, Rockville, MD 20852, under oath or affirmation under the provisions of Section 182a of the Atomic Energy Act of 1954, as amended, and 10 CFR 50.54(f). In addition, submit a copy of each response to the appropriate regional administrator.

Reasons for Information Request

Through-wall cracking of VHP nozzles and extensive degradation of the reactor coolant pressure boundary is not consistent with NRC regulatory and plant technical specifications requirements. Undetected circumferential cracking of VHP nozzles and degradation of the RPV head can pose a safety risk if permitted to progress to the point that the integrity of the reactor coolant pressure boundary is in question and the risk of a LOCA or probability of a VHP nozzle ejection increases.

This information request is necessary to permit the NRC staff to further assess plant-specific compliance with NRC's regulations. The staff will also use this information to determine the need for, and guide the development of, additional regulatory actions (e.g., generic communication, rulemaking, or orders) to address the integrity of the reactor coolant pressure boundary. Such regulatory actions could include regulatory requirements for augmented inspection programs under 10 CFR 50.55a(g)(6)(ii) to ensure that inspection practice is commensurate with the current understanding of the mechanics and likelihood of circumferential cracking and degradation phenomena. The NRC staff will review the responses to this bulletin to determine whether the PWR addressees' inspections provide reasonable assurance that existing applicable regulations are met. If concerns are identified, the NRC staff will contact the affected addressee.

Related Generic Communications

- Bulletin 2002-01, "Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity," March 18, 2002. [ADAMS Accession No. ML020770497]
- Information Notice 2002-11, "Recent Experience with Degradation of Reactor Pressure Vessel Head," March 12, 2002. [ADAMS Accession No. ML020700556]
- Bulletin 2001-01, "Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles," August 3, 2001. [ADAMS Accession No. ML012080284]
- Information Notice 2001-05, "Through-Wall Circumferential Cracking of Reactor Pressure Vessel Head Control Rod Drive Mechanism Penetration Nozzles at Oconee Nuclear Station, Unit 3," April 30, 2001. [ADAMS Accession No. ML011160588]
- Generic Letter 97-01, "Degradation of Control Rod Drive Mechanism Nozzle and Other Vessel Closure Head Penetrations," April 1, 1997.

- Information Notice 96-11, "Ingress of Demineralizer Resins Increases Potential for Stress Corrosion Cracking of Control Rod Drive Mechanism Penetrations," February 14, 1996.
- Information Notice 90-10, "Primary Water Stress Corrosion Cracking of INCONEL 600," February 23, 1990.
- Generic Letter 88-05, "Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components in PWR Plants," March 17, 1988.
- NUREG/CR-6245, "Assessment of Pressurized Water Reactor Control Rod Drive Mechanism Nozzle Cracking," October 1994.

Backfit Discussion

Under the provisions of Section 182a of the Atomic Energy Act of 1954, as amended, and 10 CFR 50.54(f), this ~~generic letter~~ bulletin transmits an information request for the purpose of verifying compliance with existing applicable regulatory requirements (see the Applicable Regulatory Requirements section of this bulletin). Specifically, the requested information will enable the NRC staff to determine whether current inspection practices for the detection of cracking in the VHP nozzles and RPV head degradation at PWR facilities provide reasonable confidence that the integrity of the reactor coolant pressure boundary is being maintained. No backfit is either intended or approved by the issuance of this bulletin and therefore, the staff has not performed a backfit analysis.

***Federal Register* Notification**

A notice of opportunity for public comment on this bulletin was not published in the *Federal Register* because the NRC staff is requesting information from PWR licensees on an expedited basis for the purpose of assessing compliance with existing applicable regulatory requirements and the need for subsequent regulatory action. This bulletin was prompted by the discovery of circumferential cracking in CRDM nozzles (above the nozzle-to-vessel head weld) from the OD to the ID and cracking in the J-groove weld metal itself, in conjunction with significant RPV head degradation. The occurrence of these two phenomena together had not previously been identified in PWRs. As the resolution of this matter progresses, the opportunity for public involvement will be provided.

Small Business Regulatory Enforcement Fairness Act

The NRC has determined that this action is not subject to the Small Business Regulatory Enforcement Fairness Act of 1996.

Paperwork Reduction Act Statement

This bulletin contains information collections that are subject to the Paperwork Reduction Act of 1995 (44 U.S.C. 3501 et seq.) These information collections were approved by the Office of Management and Budget, approval number 3150-0012, which expires July 31, 2003.

The burden to the public for these mandatory information collections is estimated to average 140 hours per response, including the time for reviewing instructions, searching existing data sources, gathering and maintaining the data needed, and completing and reviewing the information collection. Send comments regarding this burden estimate or on any other aspect of these information collections, including suggestions for reducing the burden, to the Records Management Branch (T-6 E6), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by Internet electronic mail to INFOCOLLECTS@NRC.GOV; and to the Desk Officer, Office of Information and Regulatory Affairs, NEOB-10202 (3150-0012), Office of Management and Budget, Washington, DC 20503.

Public Protection Notification

The NRC may not conduct or sponsor, and a person is not required to respond to, an information collection unless the requesting document displays a currently valid OMB control number.

If you have any questions about this matter, please contact one of the technical contacts or lead project managers listed below, or the appropriate Office of Nuclear Reactor Regulation (NRR) project manager.

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